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APPENDIX E

SUMMARY OF INEL WASTE RELEASES OR DISPOSAL, WASTE SYSTEMS IMPROVEMENTS DURING 1975 AND 1976, AND PROJECTIONS OF FUTURE WASTE AND MAJOR EXPANSION PROJECTS

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This appendix (Section 1) provides a detailed tabular summary of radioactive and nonradioactive effluent releases to the atmosphere and to the ground during 1975 and 1976. Disposal and/or storage of solid wastes is included. The environmental impact of the releases is discussed in Section 2, which is a summary of results of the extensive on- and offsite monitoring program conducted by the ERDA-ID Health Services Laboratory. Section 3 describes briefly the more significant improvements to waste management systems completed since December 1974, while Section 4 deals with projected improvements to waste management facilities in the near future. Section 5 describes ultimate disposal research programs. Section 6 discusses the projections of future wastes at INEL and Section 7 outlines the major expansion projects.

Radioactive and nonradioactive waste releases were generally less than those of 1974, the environmental impacts of which were analyzed in detail in Section III of the statement. Environmental impact of 1975 and 1976 releases from INEL operations was negligible. Significant improvements to waste management systems either completed or begun during 1975 and 1976 required expenditures of approximately $7 million, not including the New Waste Calcining Facility at the ICPP, which is projected to cost $65 million. Currently planned improvements will total more than $10 million.

1. Waste Releases/Disposal

Table E-1 presents a detailed summary of airborne radioactive effluent releases by INEL facility (area) and radionuclide during 1975 and 1976 [E-1, E-2]. The only two radionuclides released in large quantities and of significant radiological half-life (years) are krypton-85 and hydrogen-3 (tritium). Essentially all of these two radionuclides are released during nuclear fuel reprocessing and waste calcining activities at ICPP. Although the releases of strontium-90 and plutonium are small, totals for these radionuclides are shown because of their high biological significance.

Table E-2 presents a summary of liquid radioactive effluent releases by the INEL facility (area) and radionuclide during 1975 and 1976. The only two facilities contributing significantly to liquid releases are ICPP and TRA. As discussed in the statement, ICPP releases are directly to the aquifer via a disposal well while TRA liquid radioactive wastes are released to a surface leaching pond. Again, strontium-90 and plutonium totals are shown because of their biological significance.

TABLE E-1

NUCLIDE COMPOSITION (CURIES) OF AIRBORNE RADIOACTIVE WASTE FOR 1975-1976

Nuclide	ANL 1975	ANL 1976	ARA 1975	ARA 1976	CFA 1975	CFA 1976	[illegible] 1975	[illegible] 1976	NRF 1975	NRF 1976	PBF 1975	PBF 1976	TAN 1975	TAN 1976	TRA 1975	TRA 1976	Total 1975	Total 1976
Antimony-125							T	T					T[a]				<1	<1
Argon-41	185.4	178.6							5.629	T					4,879	4,835	5,070	5,014
Arsenic-76	<1																	[illegible]
Barium-139															1,132	1,283	1,132	1,283
Barium-140								T				T						T
Bromine-82	<1	<1															<1	<1
Cerium-141							T					T	T				T	T
Cerium-144							T	T					T			T	T	<1
Cesium-134				T			<1	T				T		T		T	<1	<1
Cesium-136												T						T
Cesium-137				T			T	<1				T	T	T		T	<1	<1
Cesium-138															110.7	145.8	110.7	145.8
Chromium-51												T	T	T			T	T
Cobalt-58													T			T	T	T
Cobalt-60				T			T	T		T		T	T	T		T	T	T
Europium-154							T	T									<1	T
Europium-155							T										<1	
Hafnium-181												T		T				T
Hydrogen-3(Tritium)	1.276	1.279					237.4	997.9	T	T							238.7	999.2
Iodine-131	<1			T				T	<1	T		T		T		T	<1	<1
Iodine-132										T								T
Iodine-133												T						T
Iodine-134										T								T
Iodine-135										T								T
Krypton-85	<1						24,200	31,740	<1	<1							24,200	31,740
Krypton-85m															2,475	2,867	2,475	2,867
Krypton-87	<1														10,240	12,270	10,250	12,270
Krypton-88	1.169	2.770								T					8,015	9,332	8,016	9,335
Krypton-89																		
Lanthanum-140												T						T
Manganese-54							T	T								T	<1	T
Molybdenum-99				T								T						T
Neodymium-147																T		T
Neptunium-237							T										T	
Niobium-95							<1	T				T	T				<1	<1
Nitrogen-13																		
Plutonium-238							<1	T									<1	T
Plutonium-239-240							T	T									T	T
Rubidium-88	<1	<1													100.2	115.7	100.2	117
Rubidium-89																		
Ruthenium-103							T	T				T					T	T
Ruthenium-105							T	T									<1	<1
Samarium-153												T						T
Scandium-46													T				T	
Silver-110m												T						T
Strontium-90				T			<1	<1				T	T	T		T	<1	<1
Tellurium-132												T						T
Thorium-232							T	T									T	T
Unidentified Alpha				T					T			T	T	T		T	T	T
Unidentified Beta-gamma			T	T					T	T	T	T	T	T		T	T	T
Uranium-234							T										T	
Uranium-235							T										T	
Uranium-236							T										T	
Uranium-238							T										T	
Xenon-133	338.9	277.3							T	T		T			1,019	997.5	1,358	1,275
Xenon-135	140.4	96.15													8,812	10,250	8,953	10,350
Xenon-135m															4,121	5,195	4,121	5,195
Xenon-138									T	T					16,050	21,140	16,050	21,140
Zirconium-95							T	T									T	<1
Totals	668.5	556.2	4.891(-7)[b]	1.642(-6)	0	0	24,440	32,740	6.342	6.954(-1)	1.?03(-5)	1.715(-3)	3.012(-4)	5.364(-4)	56,960	68,430	8.207(4)	1.017(5)
Volumes (meters^3)	1.977(9)[b]	2.175(9)	2.666(7)	2.679(7)	0	0	1.389(9)	1.545(9)	2.227(9)	2.218(9)	[illegible]	8.581(7)	5.529(8)	1.369(9)	8.087(8)	1.164(9)	7.082(9)	8.284(9)

[a] T = < 10^{-3} Ci.

[b] 1.977 (9) = 1.977 x 10^9

TABLE E-2

NUCLIDE COMPOSITION (CURIES) OF LIQUID RADIOACTIVE WASTE FOR 1975-1976

Nuclide	Area: ANL 1975	ANL 1976	CFA 1975	CFA 1976	ICPP 1975	ICPP 1976	NRF 1975	NRF 1976	PBF 1975	PBF 1976	TAN 1975	TAN 1976	TRA 1975	TRA 1976	Total 1975	Total 1976
Antimony-122														<1		<1
Antimony-124														1		[illegible]
Antimony-125					<1	1				T					<1	<1
Barium-140									T[a]				26.39	43.43	26.39	43.43
Carbon-14					<1	<1	T	T							<1	<1
Cerium-141									T	T			18.45	26.80	18.45	26.80
Cerium-143										<1			1.655	2.864	1.655	2.895
Cerium-144			<1	<1	<1	<1							36.67	30.72	36.85	30.78
Cesium-134			1		<1	<1				T		1	<1	<1	<1	<1
Cesium-137			<1	<1	<1	<1	T	T	T	1	T	<1	2.649	1.972	3.160	2.689
Chromium-51				<1					T	<1			1.694	2.567	1.694	2.567
Cobalt-58									T	1	T		<1	<1	<1	<1
Cobalt-60			<1	1	T	T	T	T	T	T	1	1	4.304	3.510	4.316	3.530
Europium-154					1	T									<1	T
Europium-155					T										T	
Hafnium-181										T		<1		9.963		9.965
Hydrogen-3(tritium)	<1	<1			42.70	57.48	1.001	1	T	1	<1	<1	260.2	292.90	304.6	3[illegible].10
Iodine-129						<1							T	T	T	<1
Iodine-131					1	<1			T	<1			24.72	45.35	24.72	4[illegible].00
Iodine-132													<1	<1	<1	[illegible]
Iodine-133										T			2.608	4.780	2.608	4.7[illegible]
Iodine-135													1	<1	<1	<1
Iron-59														<1		<1
Lanthanum-140						T				<1			2.734	4.823	2.784	4.826
Lutetium-177													<1		<1	
Manganese-54						T	T		T	T			<1	<1	<1	<1
Molybdenum-99										1		<1		<1		<1
Nyodymium-147													1.530	1.816	1.530	1.816
Neptunium-237					<1										<1	
Neptunium-239										<1			3.879	7.561	3.879	7.564
Niobium-95					<1	<1	T			1			6.509	11.00	6.539	11.54
Niobium-97													<1	<1	<1	<1
Plutonium-238					<1	<1									<1	<1
Plutonium-239-240					<1	<1									<1	<1
Plutonium-240																T
Rhenium-188													<1		<1	
Ruthenium-103					<1	<1			T	T			15.82	17.58	15.83	17.59
Ruthenium-106			<1		<1	<1			T				<1		<1	<1
Samarium-153										<1						<1
Scandium-46													<1	<1	<1	<1
Silver-110							T		T	T					T	T
Silver-110m								T		T						T
Sodium-24													26.90	100.80	26.90	100.80
Strontium-89						<1			T	T		T	20.55	28.55	20.55	28.57
Strontium-89-90										T						T
Strontium-90			<1	<1	<1	<1	T	T	T	T	T	<1	3.801	2.946	4.275	3.351
Strontium-91													3.514	7.756	3.514	7.756
Strontium-92													<1	<1	<1	<1
Tantalum-182													1.089	<1	1.089	<1
Technetium-99m													1.344	4.621	1.334	4.621
Tellurium-132										T			1.502	1.402	1.502	1.403
Tungsten-187													<1	<1	<1	<1
Unidentified Alpha	T	T	<1	T			T		T	T	T	T	<1	<1	<1	<1
Unidentified Beta-gamma	T	T	<1	1			T	T	T	<1	1	<1			<1	<1
Xenon-133										<1			1.265		1.265	<1
Xenon-135		<1											<1		<1	<1
Yttrium-88												T				T
Yttrium-91m															13.42	13.42
Yttrium-92													1	1.358	<1	1.358
Yttrium-93													1	1	1	1
Zinc-65														<1		<1
Zirconium-95					1	1				T			13.58	17.03	13.53	17.35
Zirconium-97													<1	<1	<1	1
Totals	1.723(-1)[b]	1.700(-1)	2.543(-1)	8.597(-2)	44.52	60.8[illegible]	1.002	4.226(-1)	8.342(-3)	1.016(-1)	4.728(-1)	1.371(-1)	2180	3255	2227	3317
Volumes (liters)	1.306(6)	3.214(6)	1.826(8)	1.599(8)	1.040(9)	1.345(9)	1.048(6)	6.451(5)	1.579(6)	1.993(6)	4.277(3)	2.2[illegible]7(7)	8.326(8)	7.583(8)	2.102(9)	2.298(9)

[a]. T = < 10^{-3} Ci.

[b] 1.732(-1) = 1.723 x 10^{-1}

Table E-3 shows the volume and activity of solid radioactive waste interred at the RWMC Subsurface Disposal Area including facility of origin during 1975 and 1976.

Table E-4 presents the volume and activity of solid radioactive wastes disposed of or stored at all INEL storage/disposal areas during 1975 and 1976.

Table E-5 lists quantities of nonradioactive wastes[E-2,E-4] by type disposed of or released during 1975-1976. Since 1974, all waste oils and solvents are shipped to TAN where they are mixed with fuel oil and burned. Prior to that time, they were used for dust control on INEL and surrounding community roads.

TABLE E-3

SOLID RADIOACTIVE WASTES DISPOSAL AT INEL DURING 1975-1976

	Volume (m^3)		Activity (Curies)	
Facility of Origin	1975	1976	1975	1976
ANL	343	483	122	425
ARA	11	7	<1	10
CFA	402	29	5	<1
ICPP	1,741	1,815	7,634	1,107
NRF	270	305	5,067	2,385
PBF	4	16	<1	<1
PER (formerly SPERT)	0	163	0	<1
RFO (Rocky Flats)	1,595	2,131	6	2
TAN	120	1,019	37	325
TRA	1,167	242	643	214,300
WMC	39	4	<1	<1
Totals	5,693	6,214	13,520	218,600

TABLE E-4

SOLID RADIOACTIVE WASTE DISPOSAL/STORAGE DURING 1975-1976

Area	Volume(m^3)		Activity (Curies)	
	1975	1976	1975	1976
RWMC Subsurface Disposal	5,693	6,214	13,520	218,600
Transuranic Storage Area	3,895	32,580	1,098	11,050
Intermediate-Level Transuranic Storage Facility	0	5	0	25
ANL Radioactive Scrap and Waste Facility	4	3	91,560	61,510
ICPP Calcined Waste Storage	66	90	1,523,000	2,396,000
SL-1 Burial Ground	0	0	0	0

TABLE E-5

SUMMARY OF NONRADIOACTIVE WASTE DISPOSAL/RELEASE AT INEL
(1975 and 1976)

Type	1975	1976
Sanitary (yd^3)[a]	24,250	19,420
Oils-Solvents (gallons)[b]	8,617	6,000
Chemical		
Liquid (lb)		
Surface	3,053,000	2,989,000
Subsurface	1,550,000	1,508,000
Solid (lb)	15	451
Airborne (lb)		
NO_2[c]	580,600	572,500
SO_2[d]	1,201,000	956,300
Particulate[d]	103,000	69,500

[a] Includes industrial trash, cafeteria garbage, wood and scrap lumber, masonry and concrete, and scrap metal.

[b] All waste oils and solvents are used by TAN as fuel oil.

[c] Approximately 92% from ICPP calciner operation.

[d] From combustion of fuel oil.

2. Environmental Impact

As in the past, the environmental monitoring program[E-5,E-6] for INEL and vicinity was conducted by the ERDA-ID Health Services Laboratory.

The results of the various monitoring activities for 1975 and 1976 indicated that radioactivity from INEL operations was indistinguishable from worldwide fallout and natural radioactivity in the region surrounding INEL. Radioactive materials and nonradioactive compounds were discharged from INEL operations, but concentrations and/or doses to onsite and offsite populations were of no health significance. Site-contributed radioactivity and nonradioactive pollutant concentrations remained far below State of Idaho and federal guidelines for safe discharge to the environment.

Air, water, and soil were monitored routinely for radioactivity at a number of onsite, perimeter, and distant locations. Milk and wheat samples were also collected routinely at both perimeter and distant locations, and the samples were analyzed for radioactivity. Penetrating radiation exposure rates (cumulated from November 1974 to November 1975 and from November 1975 to November 1976) were measured at community locations near the site boundary as well as at several distant communities. No statistical differences were noted in particulate beta and specific radionuclide measurements in the air at boundary and at distant sampling points. No offsite well water or surface water samples contained gross alpha, gross beta, or tritium activity above the detection limits of the analyses (well below the applicable health protection guides). Analyses for radioactivity in surface soil samples collected at INEL boundary and distant sampling locations indicate that INEL-contributed activity cannot be distinguished from worldwide fallout activity. Of 152 milk samples in 1975 and 156 in 1976, a total of 16 samples showed concentrations of I-131 slightly above the detection limit. Seven of these were attributable to worldwide fallout activity, two to contamination of the samples during collection or analysis, and seven to statistical fluctuations. In general, concentrations of I-131 in milk were lower in the Idaho Falls area than in many other areas of the United States. The seven samples whose levels were attributable to worldwide fallout were collected during October 1976 following the atmospheric nuclear weapons test in the People's Republic of China. Strontium-90, detected in a few milk, wheat, and potato samples during 1976, was also attributed to worldwide fallout. Penetrating radiation measurements at INEL boundary and distant locations by thermoluminescent dosimetry showed annual exposures not statistically different. This amounted to 115 mrem in 1975 and 116 mrem in 1976, all of which resulted from natural background radiation.

Airborne effluent data and a 1975 meteorologic model of the site and vicinity were used to calculate that a maximum whole body dose of 0.67 mrem from immersion in and inhalation of airborne radioactivity could have been received by a hypothetical individual if he had lived continuously for a year on the southern boundary of the site. This hypothetical dose is 0.4% of the natural background radiation of about 165 mrem per year in this area. In 1976, this maximum whole body dose was calculated to be 0.19 mrem. The reduction, compared to 1975, was due primarily to calculation refinements, not to decreased airborne effluent releases. Maximum population doses from INEL operations to the approximately 93,000 persons residing within a 50-mi radius of the center of INEL operations were calculated to be about 3.3 man-rem in 1975 and 1.2 man-rem in 1976. These hypothetical doses are 0.01 to 0.02% of the population dose from natural background radiation.

Potential dose to an individual from ingestion of meat of game animals was calculated to be less than 40 mrem/yr. This is based on the highly conservative assumption that a person consumed all the flesh of collected specimens of antelope and ducks having the highest levels of measured radioactivity. For purposes of comparison, a person in the INEL area receives approximately 140-145 mrem from cosmic and terrestrial radiation each year. An additional 20-25 mrem is received from natural radionuclides in the body.

Nonradioactive atmospheric particulates were monitored routinely in 1976 and intermittently during 1975. The INEL boundary average in 1976 was 46 $\mu g/m^3$ compared to the distant sampling point average of 83 mg/m^3. In 1975, these values ranged between 30 and 46 $\mu g/m^3$ and 60 and 69 $\mu g/m^3$, respectively. The detection limit was 35 $\mu g/m^3$. The higher values for locations away from INEL probably reflect the greater agricultural and industrial activity along the Snake River, whereas most of the particulate activity in the INEL vicinity is probably windblown dust from the desert floor. Table E-6 presents the calculated maximum concentrations of sulfur dioxide and nitrogen dioxide at the point of maximum offsite concentration, the southern INEL boundary. Standards shown are EPA national secondary ambient air quality standards.

In January 1975 INEL was designated as the nation's second National Environmental Research Park. The area was set aside as a controlled, protected outdoor laboratory where scientists from throughout the nation may study ecological changes resulting from man's activities. Research programs[E-7] currently underway involved environmental impact studies of energy related activities at INEL, the natural functioning of the ecosystem, and effects of experimental manipulation of parts of the environment. Thus far, ERDA has funded research programs in cooperation with

TABLE E-6

PRINCIPAL NONRADIOACTIVE POLLUTANT CONCENTRATIONS AT INEL (1975-1976)

	SO_2 ($\mu g/m^3$)	NO_2 ($\mu g/m^3$)
1975	0.9	0.4
1976	0.4	0.4
EPA Standard	60	100

student and staff representatives of Colorado State, Utah State, Idaho State, and Brigham Young Universities and the Universities of Idaho and Minnesota. The U. S. Fish and Wildlife Service and Idaho Department of Fish and Game have also conducted research programs at the INEL. Through 1975, 14 studies had been completed, or were ongoing, related to effects on native flora and fauna of man-made radionuclides introduced into the desert scrub biome of INEL. An additional nine basic ecological studies and surveys, primarily faunal, were either completed or underway. Receiving principal attention were antelope, waterfowl, mourning doves, rabbits, rodents, and coyotes.

3. Improvements to Waste Management Systems

Several significant improvements to INEL waste handling systems were begun and/or completed during 1975 or 1976. These are discussed briefly by facility.

A. Argonne National Laboratory - West

(1) Airborne Waste Systems

Radioactive

Experimental Breeder Reactor - II (EBR-II)

A new building to house the Sodium Component Cleanup Facility has been completed at a cost of \$462,000, although the process systems are not yet installed. Building air is exhausted through one stage of HEPA filters to a 48-ft-high stack at the rate of 10,000 ft^3/sec. This will prevent the release of a total of 24 mCi/yr of cesium-137 and sodium-24. The stack is continuously monitored for gross alpha and gross gamma.

Hot Fuel Examination Facility - South (HFEF-S)

The gaseous effluent previously discharged from the separate, unmonitored exhaust stack has been rerouted to the monitored main stack. This eliminates the use of the separate stack. Cost of the project was \$17,000.

Nonradioactive

EBR-II

A modification and cross-connection was performed which allows the use of reactor power plant steam for building heating. This reduces the amount of #2 fuel oil consumption by ∿50% per year with an attendant reduction in the amount of SO_2 discharged. Approximate cost was \$25,000.

(2) Liquid Waste System

Radioactive

EBR-II

The underground tank is no longer in use. The liquid from the component cleanup pad flows to an impervious sump from which it is automatically pumped through an aboveground pipe to the 4,000-gallon retention tank in the Sodium Component Cleanup Building. The tank is contained in an accessible concrete pit. The contents of the tank are sampled and pumped to a portable tank for transfer to the Laboratory and Office (L&O) Building waste treatment facility. The personnel decontamination liquid is collected in a tank from which it is pumped to the portable transfer tank instead of being piped underground to HFEF, then to the L&O Building. This modification was made to eliminate some untestable underground piping.

HFEF-S

Piping has been rerouted so that decontamination drains flow directly to the 1,500-gallon retention tank. A 500-gallon tank has been added to receive any overflow from the 1,500-gallon tank and the connection to the industrial waste has been eliminated. The subassembly wash has been routed to go first to the 30-gallon tank, then to the 300-gallon tank, then to the 1,500-gallon tank.

(3) Liquid Waste Processing System

The system has been modified so that radioactive waste from all facilities is passed through strainers prior to entry into the settling tanks. As the waste is passed from the settling tanks to the feed tanks, it is filtered to remove many of the solids.

The evaporator heating system has been modified so that in the event of an alarm on the installed monitor, the steam condensate is automatically diverted to the evaporator condensate tank rather than to the industrial waste.

The sludge concentration system has been modified to include a moisture separator in the ventline. Any moisture removed during the concentration process is diverted back to the settling tanks.

The modifications to ANL-W liquid waste systems were performed at a total cost of ∿$123,000.

B. Idaho Chemical Processing Plant (ICPP)

(1) Airborne Waste Systems

Radioactive

Atmospheric Protection System

Installation of the Atmospheric Protection System (APS) [E-8] described in Section II.A.3,4 was completed in July 1975. Approximate cost of the system was $3 million. A safety analysis of the potential hazards associated with the APS indicates that the system is capable of withstanding severe design basis natural phenomena (flood, tornado, and earthquake) without releasing unacceptable amounts of radioactive particulate from the filters to the environment. In-cell explosion, fire, mechanical damage, and other postulated accident situations were investigated, and from these a design basis accident (the one resulting in the most severe consequences) was postulated. This was complete release of the maximum amount of radioactive particulate collected on the 104 ventilation air HEPA filters to the atmosphere via the 250-ft-high main plant stack. Although the release of all particulate in the filters is incredible, such an event, even under the most pessimistic meteorological conditions, would not approach existing guideline limits (10 CFR 100) to personnel either on or off INEL.

Because of the lack of duplication of identical operating conditions before and after installation of the APS, a definite determination of the effectiveness of the system has not been possible to date. Preliminary estimates and approximate trends, however, indicate that a reduction factor of about 50 to 100 in particulate activity discharged from the plant stack is being realized.

The APS was designed to provide primary filtration for plant ventilation air and backup filtration for individual process cleanup systems. As a matter of ICPP policy, in the designs of such process off-gas stream decontamination systems, no credit may be taken for the added cleanup afforded by the APS.

Nonradioactive

NO_x Analyses

A photometric NO_x analyzer was installed in April 1976 to monitor oxides of nitrogen released from the 250-ft-high ICPP stack during calcination of radioactive liquid waste. Samples are taken automatically every 10 minutes and analyzed for NO, NO_2, and NO_x.

(2) Liquid Waste Systems

Radioactive Liquid Waste System Improvements

As indicated in Section IX, this project[E-9] will provide significant upgrading of existing liquid waste management systems. Construction was begun in October 1976 and will be completed in January 1978.

During the evolution of the ICPP tank farm system, there has been no significant change in tank design, type of instrumentation, or off-gas system. Also, from the time of the first expansion program (1954-1955), the same basic design has been used for the waste transfer system, including encasement, valve diversion boxes, encasement sump sampling and monitoring provisions. Items to be upgraded under the project include the following:

(a) The existing low-level radioactive waste transfer line from the Fuel Element Storage Facility, CPP-603, is presently not encased for secondary containment. This line will be abandoned and replaced with a new line in an all-welded stainless steel pipe encasement, including leak detection monitoring manways. The CPP-603 collection and pump system also will be modified for increased capacity and improved maintenance.

(b) The present concrete shielding blocks above the tank and vault-access risers are bulky and heavy, requiring operators and heavy equipment from the Central Facilities Maintenance Department for removal, a time-consuming and costly procedure. New segmented shielding blocks will be designed for removal and handling with lightweight equipment—onsite at ICPP.

(c) The vessel off-gas system for the waste storage tanks in the tank farm area is dependent on the CPP-604 vessel off-gas (VOG) control system which operates at a considerably higher negative pressure than that permissible in the tank-farm branch line to ensure proper, continuous control. Demisters also will be installed to ensure proper moisture removal from the off-gas prior to particulate filtration. Tank WM-181, which presently does not have a direct connection with the VOG system, will be connected through a new 3-in. line.

(d) Valves in four existing diversion-valve boxes are inaccessible for leak inspection and maintenance. These boxes will be modified with access manways for routine maintenance and the existing valves inspected and reconditioned as necessary. Other valves and valve clusters which presently are buried and inaccessible for maintenance will be exposed, inspected, and provided with access manways for routine maintenance.

(e) The existing provisions for sampling of valve box and pipe-encasement sumps do not permit continuous monitoring for leakage from primary waste lines into the secondary containment encasements. A radiation monitoring system will be provided for each encasement and valve box sump, with signal transmission to a central computer control and alarm system.

(f) The tank storage vaults are each provided with two or more collection sumps for leak detection. Groundwater leakage from rain and snowmelt has been experienced through the vault structure, requiring routine transfer to maintain a proper minimum level in the sumps. These sumps are presently provided with steam jets for return of sump contents to the respective tanks. Since tank wastes are processed through the Waste Calcining Facility at considerable cost, it is desirable to provide for alternate transfer to the low-level waste evaporation system (PEW). This has been accomplished by insertion of a temporary removable steam-jet assembly through sump access riser and temporary aboveground piping. Permanently installed steam jets with properly encased underground piping will be provided for routine transfer to the PEW system.

(g) The existing pneumatic bubbler-type liquid-level instrumentation for each of the 11 liquid waste storage tanks provides insufficient sensitivity for detection of level changes involving less than 1,500 gallons. Tank-leak detection has been dependent upon level buildup in the tank vault collection sumps. Level instrumentation for these sumps has not been entirely reliable and leak detection has been compromised by groundwater in-leakage through the vault roof structure (see following paragraph). A recently developed electronic resonant-frequency-type liquid-level measurement instrumentation will be installed in each tank to provide detection of leaks of only 60 gallons.

(h) Groundwater leakage into the tank vaults tends to mask out potential leakage of radioactive liquids into the vault sumps. An impervious membrane covering will be provided over the entire bermed tank-farm area to prevent groundwater and snowmelt entering the vaults.

Service Waste Diversion System

The ICPP service waste diversion system [Section II.A.3.d.(1)] was modified to allow full use of high-level tank WM-181 for intermediate-level radioactive liquid waste storage so that fuel reprocessing may continue longer before existing storage space is filled. This will allow recovery of over 100 kg of U-235, worth about $20 million, several years sooner. WM-181 has been replaced for use in the diversion system by a carbon-steel tank located above grade about 80 ft northeast of the control house (CPP-628). To date, diversions have occurred only during process equipment waste (PEW) condensate pumpouts to the injection well through the ion-exchange system. Low pH condensate passing through the ion-exchange system can release fixed activity and cause the waste to be diverted. This project was begun in August 1976 and completed in January 1977 at an approximate cost of $360,000.

(3) Solid Waste Systems

New Waste Calciner Facility (NWCF)

Construction of this major new project[E-10] at the ICPP began in September 1976 and is scheduled for completion in December 1979 at a cost of $65,000,000 based on current projections.

The NWCF will replace an existing (the world's first) plant-scale prototype Waste Calcining Facility (WCF), built in 1963. The new facility will use the same fluidized-bed calcination process proved reliable in the WCF to convert high-level radioactive liquid waste to a granular solid calcine much like smooth sand in physical appearance. The facility will also include a sophisticated decontamination area, where contaminated items can be remotely decontaminated and, to a limited extent, remotely maintained—thereby maintaining as-low-as-reasonably-achievable (ALARA) exposure limits to plant personnel. Improvement in (a) materials of construction, (b) process systems, (c) operating methods, and (d) equipment design will be incorporated into the NWCF by evaluating the latest technology and the 11 yr of successful operations and maintenance experience gained from the WCF. Furthermore, remote techniques will be developed and used throughout the facility, especially in areas where radiation fields are high and maintenance activities are most probable. The NWCF will convert high-level radioactive liquid waste from ICPP's fuel recovery processes into a solid. The calcined product will occupy only about one-eighth of the volume of the original liquid waste feed. The product will be stored in the vented stainless steel bins contained in underground concrete vaults located near the NWCF.

The NWCF will conform to ERDA's safety and ecological philosophy and guidelines and will be designed to achieve a high on-line capability, ALARA radiation exposures for onsite and offsite personnel, and the lowest practicable impact on the environment.

The NWCF will be a concrete and steel structure located north and east of the existing WCF. The capacity of the NWCF will be 60% greater than the WCF for a net feed rate of 3,000 gallons/day of liquid waste. Approximately 800,000 gallons of waste will be processed per year. The new plant will incorporate both layout and process changes that operating experience with the existing WCF has shown are needed for improved operation, maintenance, and reduced radiation exposures. All process cells will be located underground; the control room, Health Physics office, and a new decontamination facility will be located above grade.

A major improvement in the design of the NWCF will be the placement of cubicles adjacent to the process cells. The cubicles will house equipment (e.g., sample stations, valves, and flowmeters) having a history of requiring a high frequency of maintenance. The cubicles will be equipped with shielded windows and manipulators to permit as much remote decontamination and maintenance as possible.

The NWCF will have the following major advantages compared to the WCF: (1) a combination of direct and remote maintenance which will significantly decrease radiation exposure to operating and maintenance personnel; (2) increased reliability through simplicity in design and selection of construction materials compatible with existing and future wastes; (3) an improved dry and wet off-gas cleanup system which will remove a greater fraction of solids prior to the off-gas reaching the final off-gas filters; (4) overall decrease in radioactive effluent releases to the environment resulting from improved process and equipment designs; and (5) improved waste transfer and leak detection capability.

The key process step in the NWCF will be the solidification of the liquid radioactive wastes in a fluidized-bed calciner. Feed will be atomized by air into a heated bed of particles (0.3 to 0.8 mm) maintained at a bed temperature in the range 400°C - 500°C. Dissolved metals and fission products oxidize to solid oxides which coat the surfaces of bed particles; though some spray drying occurs, most of the evaporation occurs on the surfaces of particles. Calcined solids will be pneumatically transferred from the fluidized bed to underground stainless steel storage bins. Process heat will be provided by in-bed combustion (currently used in the WCF).

The effluent streams from the NWCF will include process off-gases liquid wastes, and granular solids. The process off-gases wil consist of plant ventilation air, process and calciner vessel off-gases, and solids transport air which is returned from the solids storage bins to the top of the calciner vessel. The inlet ventilation air to the process cell areas will pass through a roughing filter and at least one stage of high efficiency particulate air (HEPA) filtration. As this air exhausts, it will be routed through a prefilter and two stages of HEPA filters and out the NWCF stack. The calciner process off-gas will pass through an extensive cleanup system consisting of a high efficiency cyclone, quench tower, venturi scrubber, scrubber separator, condenser, mist eliminator, heater, ruthenium adsorbers, mist eliminator, heater, prefilter, double HEPA filters, the Atmospheric Protection System (APS), and out the ICPP stack. The process vessel off-gases will combine with the calciner off-gas cleanup system just after the ruthenium adsorbers and before the last mist eliminator. The calcined solids transport air is returned to the calciner and exhausted with the other process off-gases.

Radioactive liquid wastes generated as a result of NWCF operation will be less than that currently generated in the WCF. Process cooling water (normally nonradioactive) and overhead condensate from the evaporation of contaminated liquids are the only liquid effluents discharged to the environment from the NWCF. The process cooling water will be monitored to determine if it meets ERDA Manual 0524 standards before dischar to the existing discharge well. Decontamination solutions will be evaporated; the bottoms will be sent to radioactive waste storage (at ICPP), and the condensate will be put through an ion-exchange column for strontium removal and monitored before discharge to the existing discharge well. Transfer lines between the tank farm and the NWCF will be highly corrosion resistant and will have secondary containment to aid in detection of leakage and to protect against the accidental release of activity to the environment.

Solid wastes consist of contaminated off-gas filters and other materials (e.g., blotter paper, plastic bags, rags, mops, etc.) used in cleaning and decontamination. Essentially all of the radioactivity will be in the residual solids on the off-gas filters. Filter handling capabilities will be provided to allow about 90% of the filter activity to be removed and recycled to the calciner before the filters are transported to the RWMC for disposal.

The granular product from the NWCF will be pneumatically transported in double-encasement stainless steel lines to the stainless steel solids storage bins, which are enclosed in a concrete vault. The transport system and solids storage facility will be seismically hardened against any credible

INEL earthquake. The possibility of solids escaping this system is remote, since the solids transport and return air lines are maintained at negative pressure.

Solidification of wastes generated during fuel reprocessing will significantly decrease the potential for release of radioactivity to the environment. Operation of the NWCF to convert liquid wastes to an immobile solid will result in discharge of off-gas and contaminated water within ERDA Manual 0524 standards, and shipment of contaminated solids to the INEL RWMC. Even when processing the most radioactive waste through the NWCF, offsite airborne radionuclide concentrations will be less than 0.2% of guideline (RCG) concentrations. Releases of chemical pollutants (e.g., NO_x, CO) to the atmosphere will result in offsite concentrations that are less than 0.6% of those permitted by State of Idaho air quality standards.

Fourth Set of Calcined Storage Bins

Construction began in April 1976 to provide an additional 17,000 ft^3 of storage for calcined solid wastes. The project is scheduled for completion in October 1977 at a cost of about $2.3 million. Design provides double containment of the calcined solids with a set of stainless steel bins enclosed in a reinforced concrete vault. Decay heat will be removed by conduction of the heat through the vault walls to the surrounding air and soil. Supplementary cooling will also be provided during the first few years of bin use by natural convection of cooling air through the vault. The cooling air discharging from the vault will be monitored by a system that will automatically shut off the cooling air flow if radioactive particulates are detected in it. In this case, a filter could be added and the circulation of cooling air restored if necessary before sintering of the solids occurred; the solids temperature would increase at a slow rate upon vault isolation. The vault will be equipped with a sump to collect any water that might leak in, a liquid detector, and a sump jet for water removal. The facility will be designed to resist an earthquake with a bedrock acceleration of 0.33 g and a (plank) missile-carrying tornado with a maximum wind speed of 175 mph. The general radiation field will be limited to less than 0.5 mrem/hr at the vault surface.

The double-containment design of the facility will prevent the release of radionuclides to the underlying aquifer if either a bin or the vault should fail. There would be a slight chance of release of radioactive particulates to the atmosphere as a result of a bin leak. The Design Basis Accident (DBA), which has a very low probability of actually occurring, postulates the spill of solids from an eroded fill line into the vault coupled with failure of the radiation monitor. The maximum calculated radiation dose at the nearest INEL boundary as a result of the postulated DBA is 40 mrem. There is essentially no possibility of an accidental explosion.

C . Test Reactor Area (TRA)

(1) Liquid Waste Systems

Radioactive

A project was completed in September 1976 which segregates non-radioactive liquid wastes from low-level radioactive waste discharges before they are mixed. This prevents contamination of some 41 million gallons/yr of aqueous waste which have heretofore been discharged to the TRA leaching pond. It also allows recycle of ATR primary coolant system (PCS) instrument purges back to the PCS, saving another 2 million gallon/yr discharge to the leaching pond.

A similar project for the Engineering Test Reactor (ETR) is under construction and is expected to be completed in February 1977.

A project is underway which will allow a portion of the Materials Test Reactor (MTR) canal to be drained, thus eliminating a 30-million-gallon/yr low-level discharge to the TRA leaching pond. It is scheduled for completion in 1977.

4. Projected Improvements to Waste Management Systems

A number of significant improvements to INEL waste handling systems are planned for the near future. These are discussed briefly by INEL facility for which the improvement is planned:

A. Argonne National Laboratory-West

(1) Airborne Waste Systems

Radioactive

EBR-II

The Cover Gas Cleanup System (CGCS) is a cryogenic distillation system for removing the radioactive gases xenon and krypton from the argon cover gas. The gases will be frozen and retained for decay. Only Kr-85 will remain, and is expected to accumulate at the rate of 10 Ci/yr in the system. Startup is expected in April 1977. Cost of the project was $2,180,000 and is expected to make EBR-II a near-zero release facility for gaseous activity.

The Sodium Component Cleanup Facility will have a small components cleaning system installed in 1977. This system will use two-stage HEPA filters and will eliminate the use of the outside cleanup pad. This will result in a cost of ∿$200,000.

L&O

The ventilation system of the L&O complex is being modified at a cost of $270,000 to consolidate the various exhaust points into two central L&O exhaust stacks. This will allow better filtration and control of L&O exhaust as well as removing the L&O exhaust from the main stack.

(2) Liquid Waste Systems

Radioactive

EBR-II

Installation of the Sodium Component Cleanup Facility's (SCCF) small component cleanup system will eliminate the use of the pad, thus eliminating the use of the sump and aboveground line leading to the SCCF retention tank. The cost of this modification is discussed under Airborne Waste Systems.

HFEF-S

Modifications include filters and ion-exchangers in the lines leading to the 1,500-gallon retention tank. Ability to recirculate the 1,500-gallon tank through filters and ion exchangers prior to transfer to L&O will also be installed at that time. The cost of these modifications is ∿$213,000.

HFEF-N

Modification includes filters to remove radioactivity closer to the orgin. These modifications will cost ∿$100,000.

(3) Solid Waste Systems

Radioactive

RSWF

The volume and radioactivity in the RSWF will be reduced by the removal of from 30 to 88 liners containing non-TRU waste. These liners will be sent to the RWMC for disposal. This would be performed at a cost of ∿$15,000.

B. Central Facilities Area (CFA)

Radioactive

A new laundry decontamination facility has been projected for completion at CFA during 1979 at a cost of about $1.6 million. The new facility will provide for improved control of contamination by proper building zoning, control of air flow between zones, improvement of the dryer air filtering system, use of

hospital type pass-through washers, and modern, sensitive detection instrumentation. Reduced solid, liquid, and airborne wastes will be generated compared to the wastes generated at the existing facility.
A new flocculating treatment of wash water will reduce the amount of radioactivity released to the settling basin, resulting in lower-level contamination in sludge which is transferred to the RWMC.

C. Idaho Chemical Processing Plant (ICPP)

Airborne Waste Systems

Radioactive

X-Cell Modifications

Modifications to this laboratory will upgrade or replace a number of existing facilities to allow safe handling of high specific activity alpha emitters. Included in these improvements will be rerouting of the ventilation air from gloveboxes and hoods to the Atmospheric Protection System instead of exhausting from the roof of CPP-601. Although the existing exhaust system is equipped with HEPA filters, the upgraded system will have improved bagout and waste disposal features. It will be equipped with appropriate flow, fire, and pressure indicators and controls. Duct work will be of stainless steel, and PVC and neoprene components will be replaced for fire safety. HEPA filters will be designed to allow pressure drop measurements and provisions will be made for DOP testing. Cost of the project is estimated at $340,000 and is to be completed in 1978.

CPP-602 Laboratory Filter Boxes

Filter boxes used to house HEPA filters on the radiochemical laboratory hood exhaust stream from building CPP-602 will be replaced during 1978. Replacement of these "homemade" boxes with a commercial type featuring an improved filter-charging method will significantly upgrade the sealing of leaks around the filter seats and provide a safer method of handling contaminated filters. Estimated cost of the project is $329,000.

CPP-637 Chemistry Laboratories Upgrade

HEPA filters will be added to the exhausts of laboratory hoods in which radioactive materials are used as a part of this project scheduled for completion in 1978. Currently, these hoods exhaust directly to the roof of laboratory building CPP-637 without filtration. Cost of the total project is estimated at $490,000.

D. Power Burst Facility (PBF)

Airborne Waste Systems

Radioactive

Relocation of the PBF stack monitor is scheduled to be completed by April 1977. High radiation background during high-power level operation of the reactor interfered with reliable operation of the monitor, making it necessary to relocate the instrument to a lower background area. Release information has been obtained by analysis of effluent particulate and charcoal filters following each test; however, real-time monitoring of releases during high-power operation is considered necessary.

Liquid Waste Systems

Scheduled for completion in 1979 is an evaporation pond to replace the corrosive waste disposal well. A collection sump will be provided and furnished with a pump, plus an installed spare pump, discharging to a lined, surface evaporation pond. Effluent from the PBF corrosive waste system will be drained by gravity flow to the sump. Level sensors in the sump will activate the primary pump which discharges through 3-in. PVC piping to the evaporation pond. The spare pump will be activated by a high-level detector switch should the first pump fail. A section of the existing pipe to the corrosive waste well will be used as an overflow pipe from the sump to allow waste water flow into the existing corrosive waste well should both pumps fail. High- and low-level alarms for both pumps will be installed indicating failures to start or stop.

The evaporation pond will be 140 feet square at the bottom and 4 1/2 ft deep. It will be lined with a polymer material and covered with fine sand to prevent puncture. An 8-ft-high cyclone fence will exclude animals and blowing weeds.

E. Test Reactor Area (TRA)

Radioactive

Effluent Monitoring and Sample Analysis

A program to provide significant improvement of monitoring of TRA waste discharges is underway. Highly sensitive Ge(Li) detectors will be installed at airborne and liquid effluent release points. Data will be fed to a computer and both qualitative (radioisotopic distributions) and quantitative information will be available on a real-time basis. Readouts in the control room will allow close control of plant operation with respect to waste effluent. The Advanced Test Reactor (ATR) stack monitor, ATR primary water monitor, and TRA liquid effluent monitor are expected to be included in this system.

TRA Water Recycle and Pollution Control

A long-term (several years) project is planned to drastically reduce the amount and radioactivity level of liquid effluents released to the environment. Phase I of the three-phase project is scheduled for completion in December 1977. Waste monitoring and sampling capabilities will be improved by providing direct measurement and continuous proportional sampling of wastes released to the TRA leaching pond. In the past flows have been estimated based on pump capacities. ATR canal water will be recycled to eliminate an annual low-level radioactive discharge of over 26 million gallons of waste water to the leaching pond. Intermediate-level liquid waste storage capacity will be increased by approximately 100,000 gallons to handle the higher volume of these wastes generated by the various environmental pollution measures. Intermediate-level wastes will be transported to the ICPP for volume reduction (evaporation), calcination, and storage as calcined solid radioactive waste. Scheduled for completion in October 1978 is the installation of demineralizers on ATR low-level radioactive waste streams which will effect an order of magnitude reduction in released activity.

Phase 2 of this project, scheduled for completion in 1982, will provide for recycle of ATR primary coolant systems water through the gland seals eliminating a low-level radioactive discharge to the leaching pond of about 17.5 million gallons/yr. Engineering Test Reactor (ETR) canal water will also be recycled to eliminate a 10 million gallon/yr discharge of contaminated (low-level) water to the leaching pond. Demineralizers and filters on ETR low-level radioactive waste streams will result in a reduction of discharged radioactivity. The existing cold waste disposal well is to be replaced by a surface pond to eliminate a direct discharge of some 400 million gallons/yr of heated (above aquifer temperature) and treated water directly to the aquifer (mostly cooling tower blowdown).

Phase 3, which is still in the preconceptual stage of planning, would, by about 1984, essentially eliminate all radioactive liquid waste discharges to the environment by total recycle of low-level liquid wastes.

5. Ultimate Disposal Research and Development Program

Research and development is being funded at INEL to determine ultimate waste disposal methods. The present identified research projects include the following.

A. Fluid Bed Calcination and Post Treatment of Commercial High-Level Wastes

This program consists of technology development leading to the further conversion of calcined waste to a highly stable waste form. Included in the program are development studies to evaluate and demonstrate the application of fluidized-bed technology to post calcination treatment of commercial high-level wastes. Simulated wastes will be used. The work will require design, fabrication, and cold-testing of plant-scale prototype equipment. The program has been ongoing since 1974.

B. Long-Term Management of ICPP High-Level Wastes

This program will result in the design of a system to retrieve calcined solid wastes from interim storage at ICPP and to convert them to more stable solid forms. Under investigation are (1) methods to separate actinides from retrieved high-level calcined wastes, (2) pelletization of the wastes for improved onsite storage, and (3) vitrification for safer transport and repository storage. The program has been underway since 1974.

C. Molten Salt Combustor

A demonstration plant will be established at INEL to determine the feasibility of reducing the volume of extremely low radioactive (beta-gamma) solid wastes with the use of a molten salt combustor. Performance testing will be accomplished during 1978 using low-level wastes generated at INEL. Some followup research and development effort, relating to the potential use of this combustor for alpha contaminated wastes, will also be initiated.

D. Transuranic Waste Management

Approximately 2 million ft^3 of transuranic wastes were buried at INEL from 1954 through November of 1970. This program will define the most suitable methods of handling this waste; from retrieval through processing to storage pending shipment to a federal repository. Studies will be performed and included in appropriate documents describing alternative options, retrieval options, and environmental consequences.

6. Projections of Future Wastes

Any attempt to predict future waste generation at INEL involves a number of uncertainties and assumptions. In this section, projections through 1981 are made using volumes of waste from 1971 through 1976 as a base. No major change in waste generation, either increase or decrease, is currently anticipated except as outlined. Should situations arise leading to substantial changes, environmental assessments will be prepared.

The overall philosophy of the waste management program at INEL is to reduce the environmental impact from radioactive waste as much as is reasonably achievable. Planned programs in each plant area are being implemented. These can affect each type of waste generation in a significant manner. A discussion of each is provided below.

A. Airborne Wastes

Based upon current programmatic expectation, it is projected that a decrease in airborne particulate radioactive waste will occur in the near future. The major element that will account for this decrease will be at ICPP, which presently accounts for 20% to 30% of the volume of airborne radioactive releases at INEL. The Atmospheric Protection System (described in Section II.A.3.a) reduces particulate releases from this facility by an estimated factor of 50 to 100. However, it will not reduce the krypton-85, the major radioisotope, released from fuel processing. Present releases do nonetheless meet current standards. EPA has proposed standards applicable to processing of commercial spent fuel which, if adopted by ERDA, would require rather drastic reductions in ICPP releases and would require installation of an efficient system for trapping and storing krypton-85. Projected krypton-85 releases through FY 1981 are as follows, assuming no more strict release standards apply:

FY-77	FY-78	FY-79	FY-80	FY-81
99,000 Ci	10,000 Ci	44,000 Ci	35,000 Ci	82,000 Ci

Other facilities at INEL have plans for decreasing airborne particulate emissions during the next few years. However, additional increases may result from new programs. One such program is the proposed Safety Research Experiment Facilities (SAREF) Project. This facility, proposed for construction near EBR-II, is scheduled for operation by the end of CY-1982. The SAREF project is in support of the liquid-metal fast-breeder reactor. A draft environmental statement (ERDA-1552-D) has been written for the facility. The projected radioactive wastes from this facility are mainly airborne effluents and approximately 3,000 Curies/year would be released to the atmosphere.

Taking the above variables into consideration, it is anticipated that only a 1 to 3% decrease per year might be expected in total over-all airborne contaminants released to the atmosphere at INEL due to the fact that krypton-85 is the major airborne contaminant.

B. Liquids

As with airborne releases, there are some uncertainties encountered in projecting future releases of low-level waste liquids to the environment. Nonetheless, planned major improvements are expected to reduce liquid waste releases. At the Test Reactor Area (TRA), a multimillion dollar phased program has begun to reduce the volumes, as well as the radioactive contents of low-level waste liquids, now discharged to the environment via seepage ponds. Since TRA and ICPP together are the major producers ($>$ 99%) of low-level radioactive liquids discharged to the environment, these TRA improvements will have a major effect on total INEL releases.

At the end of the second phase of this TRA waste water recycle program, volume reduction by a factor of at least 10 is expected and a reduction by a factor of about 34 is anticipated for radioactive contents. The second phase is expected to be completed in about three years. The third phase of the program now envisioned would completely eliminate liquid waste discharges to the seepage ponds at TRA.

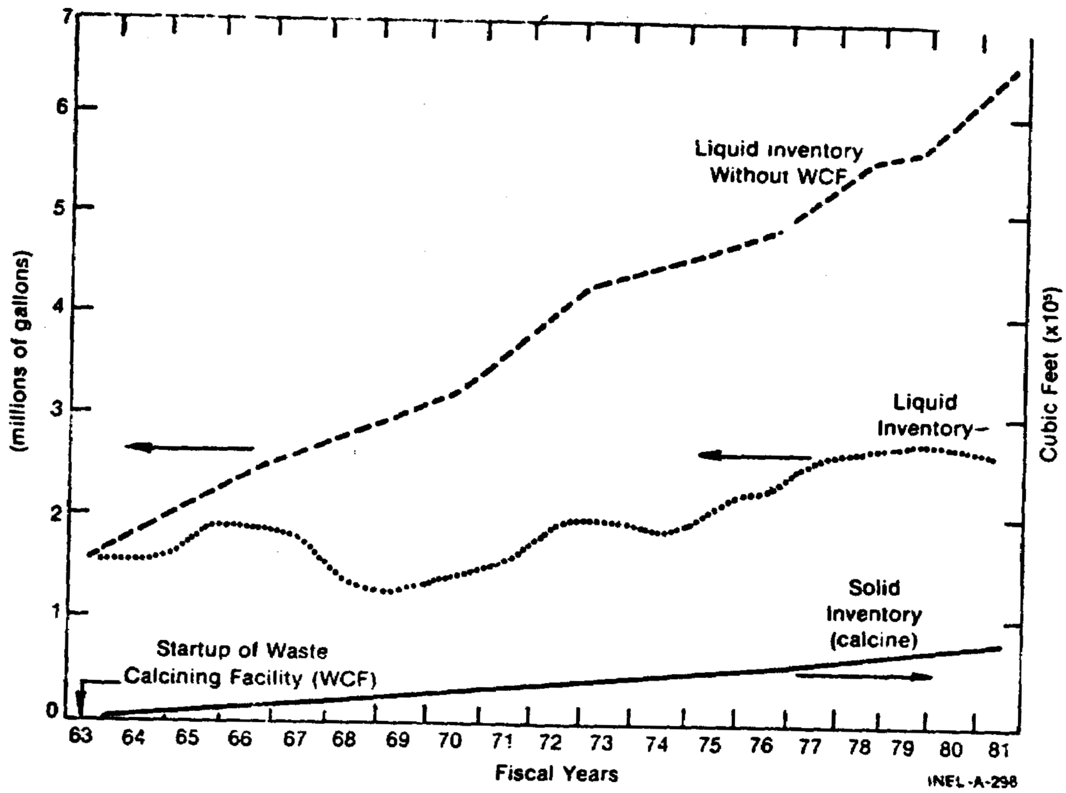
Based upon the assumptions that the TRA recycle program will meet or exceed design performance, an overall volume reduction in liquid wastes discharged to the environment at INEL can be projected by 1983. This reduction amounts to about 5% per year averaged over the period. A relatively larger reduction is expected over the early years of this period with smaller annual decreases expected as final parts of the TRA recycle program are installed. The program is expected to reduce the radioisotope released to essentially zero concentration.

There are currently no plans to reduce the volumes of liquid waste discharged at ICPP. The radioactive liquid discharged to the well meets ERDA standards for release to uncontrolled areas.

C. Solids

Although volume reduction programs are in progress or in design stages (e.g., Molten Salt Combustion Facility), the volume of solids disposed will show an increase during the period, should programs now planned be completed. The major reason for projecting increased solid waste volumes are the plans for decontaminating and decommissioning (D&D) a number of deactivated facilities. These facilities include many noted in Section II.A.12 of the statement.

At ICPP the calcination of high-level liquid wastes will produce an increasingly larger inventory of calcined solids as the liquid inventory is decreased. The chart below shows how liquid waste inventory is decreasing by operations of the waste calcining facility.



If the D&D programs progress as currently planned, over 500,000 ft^3 of solid waste will require storage or disposal. This is nearly 10% of the volume of all solid radioactive waste handled at the Radioactive Waste Management Complex since its inception in 1952. The challenge of reducing this expected large volume of solid waste is expected to produce some improvements in D&D operating procedures. Nevertheless, the amounts of waste from D&D will obviously be much greater than present annual rates of disposal of solids.

About a 20% increase in volume of solid wastes handled over the period can be expected as a result of the D&D activity. However, assuming superior volume reduction techniques can be developed, the increase could conceivably be held as low as 5 to 10%, based principally upon the successful utilization of the Molten Salt Combustor by 1980.

With respect to solid transuranic waste stored at the RWMC, it is projected that the volume of this waste handled over the period will remain relatively constant.

7. Major Expansion and Upgrading Projects

A. Fluorinel and Storage Facility

A major new facility is proposed for the ICPP site and is currently under design. The facility will provide new fuel handling and storage facilities and a new headend system for reprocessing a variety of irradiated nuclear fuels which cannot be processed with existing equipment, and those fuels presently being processed by the existing zirconium and the existing zirconium headend system. The facility will be capable of receiving, and storing for up to 10 years, fuels contained in the existing storage facility and those fuels expected to be received during the following 20 years. The facility will be capable of dissolving some of the presently stored fuels and most of the fuels expected to be received in the future. The remaining fuels will be dissolved by existing headend systems.

The fuel storage facility will provide storage and fuel element preparation capabilities immediately adjacent to the fuel dissolution area. The transfer of fuel elements from the storage pool to the fuel cutting and preparation area and thence to the Fluorinel headend will occur along shielded, controlled-airflow paths. This transfer method will greatly reduce the potential spread of radioactivity during the transfer of fuel from storage to processing.

The storage portion of the facility will provide approximately 1,800 fuel storage positions. This storage area will replace the existing 25-year-old storage facility (CPP-603), which has inadequate cask and fuel receiving capabilities and a history of increasing personnel exposure, contamination control problems, and maintenance effort.

The facility ventilation and filtration system will greatly reduce the potential for occupational exposure to contamination compared to the present fuel storage facility (CPP-603) which is not equipped with a ventilation system. The continuous airflow provided by the new facility's ventilation and filtration system will remove airborne radioactive materials from the building and environment and prevent contamination buildup.

The pool water chemistry control system will also reduce potential personnel exposure compared to the present facility. The CPP-603 water chemistry control system, due to various problems, has allowed buildup of radioisotopes in the pool water. This buildup has created a potential direct dose problem as well as a potential airborne contamination problem. The new facility's water chemistry control system will be a system of parallel pumps, filters, and ion exchangers designed to provide a pool water turnover rate for all pools of about once every two days. The system will be designed to maintain total beta activity in the pools at $<$ 1 nanocurie/ml.

The Fluorinel dissolution portion of the facility will provide a new headend dissolution system. The fuel will be dissolved with hot (60-70°C) acids. Following removal of the fuel cladding and dissolution of the fuel, the product will be airlifted with nitrogen or jetted with steam to complexing tanks. All solutions collected in the complexing tank will be chemically adjusted with aluminum nitrate to improve stability, to complex fluoride and to improve separation during solvent extraction. The complexing tank solutions will then be heated to about 80°C, using a closed loop system, until samples indicate that dissolution of uranium is complete. Undissolved solids will be removed from the product liquid by filtration. The filtered product liquid will be transferred to an existing solvent extraction system in which the uranium will be separated and purified. For the Fluorinel fuels, the extraction processes use tributyl phosphate (TBP) and Hexone (Methyl Isobutyl Ketone) as the solvents to recover greater than 99.9% of the uranium in the aqueous feeds.

The facility will be located inside the security-fenced ICPP area. It will enclose approximately 7,800 sq mi of floor area. The facility will be divided into the following functional areas:

1. A truck, railcar, and straddle carrier receiving bay that will function as an airlock, to prevent disturbing the ventilation and contamination control in the main facility, during vehicle entry and exit.

2. A cask receiving and decontamination area where vehicles can be positioned for cask transfer. The cask decontamination area will be separated from the rest of this area by partitions and directed airflow to provide contamination control.

3. A shielded fuel unloading pool for underwater remote cask loading/unloading, remote fuel inspection, remote canning of any fuel found to be leaking, and transfer to the storage basin via a transfer canal.

4. A fuel storage pool area, containing several separate stainless-steel-lined concrete pools.

5. A fuel cutting and preparation area, where fuel can be received from storage, possibly cut or otherwise prepared for processing, and stored under water until transfer to the Fluorinel or other headend. A shielded underwater transfer passage will connect this area to the dissolver charging cave. Fuel element cooling will not be necessary during dry transfer, since the decay heat content will be low as a result of the long cooling period before processing.

6. The Fluorinel process cell, hot cell and process make-up (PM) areas. Fuel elements will be transferred from the fuel cutting and preparation area, through the hot cell and into the dissolvers located in the process cell. Chemicals necessary for the dissolution will be fed into the process cell from tanks located in the PM Area. The hot cell and process cell will be shielded such that radiation doses outside the cells will be as low as reasonably achievable.

7. Operating areas, including control rooms and observation corridor, will be provided for the control and observation of all unloading, loading, and transfer of fuel elements, and for the monitoring and regulation of all controlled facility parameters.

8. Areas for offices, storage, restrooms, heating and ventilation, and emergency power generation.

The facility will be capable of receiving casks by truck and straddle carrier, and will be constructed to provide for future addition of rail-car receiving capability. The casks will be transferred to and from the vehicles and positioned to allow the under water loading/unloading of fuel in the fuel handling and transfer cell. The fuel will be inspected, canned if necessary, cut if required, and stored until its transfer to the processing cell. The fuel storage area will contain fuel storage pools with fuel storage racks for underwater fuel storage, a containment building to enclose the fuel storage and handling areas, a water chemistry control system, heat exchangers to cool the basin water, and fuel handling equipment.

The effluent treatment system for the facility will be designed and operated to assure that all releases to the environment will comply with the guidelines of ERDA Manual Chapter 0524. Ventilation exhaust air from the various potentially contaminated areas of the facility will be filtered through pre-filters and multiple high-efficiency particulate air (HEPA) filters prior to its release to the atmosphere through a new 50-meter-high facility stack. The overall effect of the ventilation and filtration system will be a reduction in particulate effluent concentrations by a factor of at least 10^7 relative to concentrations without the system and a reduction in particulate concentrations at the site boundary by a factor of at least 10^{12} relative to initial in-plant concentrations. Liquid wastes will be contained and transferred to the appropriate existing waste handling system. Confinement and safety-related systems will be designed to ensure that releases resulting from design basis accidents are well within the limits specified in 10 CFR 100.

The beneficial and adverse impacts on the environment from construction, operation, and site restoration are described below.

(a) Construction. The construction of the facility will require excavation, grading, and other earth-moving operations within the present ICPP boundary. These activities will create some dust and noise for short periods of time, but should have no long-term effect on or off the site. The installation of such roads and walkways as are necessary will not appreciably increase the total paved area at ICPP.

No radioactive dirt will be encountered during excavation at the building site. Minor amounts of low-level radioactive dirt may be encountered during excavation of trenches for pipes connecting the facility to existing ICPP systems. Total amounts of low-level radioactive dirt which may be encountered will not exceed a few cubic meters. If encountered, low-level radioactive dirt will be handled by the existing ICPP solid waste management system, and will be transferred to the INEL Radioactive Waste Management Complex for disposal.

Since the building will be located inside the ICPP security fence, there will be no wildlife displaced as a result of contruction of the facility.

There will be a permanent commitment of construction material such as concrete, steel, and other structural materials.

(b) Operation. The probable environmental impacts from the operation of the facility are discussed below. The impacts are divided, for clarity, into Fluorinel effluents, storage facility effluents, and overall facility environmental impacts other than effluents.

(1) Fluorinel Effluents. Since the total throughput rate through the Fluorinel headend will be higher and the fuel element cooling times shorter at the time of dissolution, particularly in the mid-1980's, than the majority of fuels processed in recent years, the total radioactivity in the process will increase. Radioactive liquids and solids will be processed by existing ICPP waste handling systems and will not be released to the environment. Airborne releases will be through the facility off-gas and ventilation and filtration systems. The use of prefilters and multiple-stage HEPA filters will result in reduced releases of particulate radioactive material compared to present releases. Annual releases of noble gases, I-129, and H-3 will, due to the increased throughput rate, be higher than present releases.

The potential impact on the environment of radioactive airborne, liquid and solid effluents resulting from the operation of the Fluorinel process will be negligible in the vicinity of the ICPP and beyond. These effluents are discussed below.

(i) Airborne. ^{85}Kr and ^{3}H comprise $>99.99\%$ of the radioactivity which will be released to the environment. The Fluorinel headend process design will incorporate a particulate removal system to assure that particulate removal system to assure that particulate quantities released to the environment are not increased. The facility ventilation and filtration system will further reduce particulate releases to the environment.

Table E-6 compares calculated Fluorinel process airborne releases of isotopes of interest for 1982 through 1987 with actual ICPP total releases for 1974 through 1976 (S. S. White, INEL Radioactive Waste Management Information for 1976, IDO-100055, 76(1976). Iodine-129 values are projected, based on the concentration of that nuclide in the fuel at dissolution times and assume that all this gas will be released either during dissolution or subsequent calcination. Particulate values are projected, based on the concentration of these nuclides in the fuel at dissolution time, and assume a release fraction from the Fluorinel process of 1×10^{-3} and a particulate DF (through Fluorinel headend and facility ventilation and filtration system pre-filters and multiple HEPA filters) of 10^7. Concentrations of all other radioisotopes at the INEL boundary are $> 3 \times 10^{-17}$ μCi/ml and are, therefore, well below ERDAM 0524 RCG's for uncontrolled areas.

The average size boundary concentrations for ^{85}Kr, ^{3}H and ^{129}I, in percent of ERDAM 0524 uncontrolled area RCG's, for 1974 and 1984 (the projected peak throughput year for the Fluorinel process) are given in Table E-7. The 1984 values listed in Table E-7 were calculated assuming (1) the Fluorinel process operates for 200 days per year at its design dissolution rate of 6.7 Kg ^{235}U per day; (2) fuel element dissolved is the design basis fuel element (DBFE); (3) the DBFR has been out of the reactor for one year; (4) radioisotope release fractions (to off-gas system) are 1.0 for Xe, ^{3}H, and I, and 0.001 for particulates; (5) ventilation system decontamination factors of 1.0 for ^{3}H, Rn, Kr, Xe, and I, and 10^7 for particulates; (6) release occurs at a constant rate over the 200 day "processing year"; (7) stack discharge total for 200 day operation is 8.16×10^{14} cm^3; (8) discharge is through the 50-meter-tall facility stack; (9) the distance to the INEL boundary is 1.3×10^4 meters; and (1) average site meteorological conditions exist with a resultant atmospheric dilution factor at the site boundary of 2.16×10^{-5}. Isotopes not listed in Table 2 (including all isotopes of plutonium) have concentrations at the INEL boundary of $< 3 \times 10^{-5}$ percent of ERDAM 0524 uncontrolled area RCG's. These calculations assume the complete release of krypton and xenon gases, although a portion of these gases will be recovered at the Idaho Rare Gas Plant.

TABLE E-7

CURIES OF KRYPTON-85, TRITIUM, IODINE-129, AND OTHER RADIONUCLIDES RELEASED TO ATMOSPHERE

	Total ICPP			Fluorinel Process (Estimated)											
Year	1974	1975	1976	1982		1983		1984		1985		1986		1987	
Decay Time (yrs)	4 - 10	4 - 10	4 - 10	4	1	4	1	4	1	4	1	4	1	4	1
Nuclide															
^{3}H	6.0(3)[a]	2.3(2)	1.0(3)	6.1(3)	7.2(3)	3.1(4)	3.7(4)	3.2(4)	3.8(4)	2.6(4)	3.1(4)	1.2(4)	1.4(4)	1.2(4)	1.4(4)
^{85}Kr	2.5(5)	3.8(4)	3.2(4)	1.7(5)	2.1(5)	9.1(5)	1.1(6)	9.1(5)	1.1(6)	7.7(5)	9.3(5)	3.4(5)	4.2(5)	3.3(5)	4.0(5)
^{90}Sr	3.2	2.4(-1)	5.0(-2)	1.5(-4)	1.6(-4)	7.6(-4)	8.2(-4)	7.6(-4)	8.2(-4)	6.4(-4)	6.9(-4)	2.9(-4)	3.1(-4)	2.8(-4)	3.0(-4)
^{106}Ru ^{106}Rh	3.8	7.9(-1)	1.0(-1)	5.9(-6)	4.7(-5)	3.0(-5)	2.4(-4)	3.0(-5)	2.4(-4)	2.7(-5)	2.1(-4)	1.2(-5)	9.3(-5)	1.1(-5)	8.9(-5)
^{129}I	NA[b]	NA	NA	5.0(-1)	5.0(-1)	2.5	2.5	2.5	2.5	2.2	2.2	9.6(-1)	9.6(-1)	9.1(-1)	9.1(-1)
^{137}Cs	6.7	6.0(-1)	1.1(-1)	1.5(-4)	1.6(-4)	7.7(-4)	8.3(-4)	7.7(-4)	8.3(-4)	6.5(-4)	7.0(-4)	3.0(-4)	3.2(-4)	2.8(-4)	3.0(-4)
^{238}Pu	4.0(-3)	2.0(-3)	1.0(-3)	6.4(-8)	6.6(-8)	3.4(-7)	3.5(-7)	3.4(-7)	3.5(-7)	2.8(-7)	2.9(-7)	1.3(-7)	1.3(-7)	1.2(-7)	1.3(-7)
^{239}Pu + ^{240}Pu	3.0(-4)	3.0(-4)	4.0(-4)	2.4(-8)	2.4(-8)	1.3(-7)	1.3(-7)	1.3(-7)	1.3(-7)	1.1(-7)	1.1(-7)	4.8(-8)	4.8(-8)	4.6(-8)	4.6(-8)

(a) 6.0(3) means 6.0×10^{3}
(b) NA means no analysis

TABLE E-8

BOUNDARY CONCENTRATION COMPARISON
1974 ICPP ACTUAL VS 1984 FLUORINEL ESTIMATED

	% of ERDAM 0524 Uncontrolled Area RCG	
Nuclide	1974	1984
^{85}Kr	0.5	2
^{3}H	Negligible	0.1
^{129}I	Negligible	0.08

For 1984, this would result in an estimated annual whole-body dose commitment to an individual at the nearest point on the site boundary of 0.6 man-rem/year from operation of the Fluorinel process, which is 0.12% of the ERDAM 0524 Radiation Protection Standard for individuals in uncontrolled areas and is equivalent to about 0.4% of the annual exposure one would receive from the natural radioactivity background at INEL. Thus, the anticipated Fluorinel process airborne releases are expected to have a negligible effect on the environment.

(ii) Liquid. Releases of liquids from the Fluorinel headend will be to existing ICPP liquid waste management systems or to the ICPP Solvent Extraction System.

Liquid wastes from rinsing and decontamination operations will be transferred to the ICPP process equipment waste (PEW) system. These intermediate-level wastes will be stored in the PEW system storage tanks until processed through the PEW evaporator. Concentrated PEW evaporator bottoms will be transferred to the ICPP tank farm and processed as high-level waste. Evaporator overheads condensate will normally be passed through an ion-exchange system, monitored for radioactivity and sampled for chemical content, and discharged to the disposal well along with the ICPP service waste.

The Fluorinel headend product, containing all the fuel element radioisotope inventory except that released as airborne effluent, will be transferred to the existing ICPP solvent extraction system. The solvent extraction system will discharge the high-level waste, as first-cycle raffinate, to the tank farm. This high-level waste will be held in the high-level waste tanks until it is transferred to the New Waste Calcining Facility (NWCF) for solidification and storage. The high-level waste system, with the addition of the NWCF prior to Fluorinel process operation, will have sufficient capacity to handle the liquid waste generated by the Fluorinel process. Airborne releases to the environment resulting from NWCF operation, including calcination of Fluorinel process high-level waste, will result in maximum concentrations at the site boundary of 0.2% of ERDAM 0524 uncontrolled area RCG's, with a resultant annual whole body dose of 0.003 man-rem. These releases are considered to be environmentally insignificant (NWCF Environmental Statement, WASH-1531 (June 1974)).

The only liquid waste from the Fluorinel headend that will be discharged directly to the environment is the PEW overheads condensate. This liquid is monitored for radioactivity and sampled for chemical content prior to discharge to insure it does not contain radioactive or chemical contamination. The service waste discharge, of which PEW overheads condensate is a part, occurs at ambient temperature and, therefore, has no thermal impact on the environment. Thus, there will be no impact on the environment from Fluorinel headend liquid waste.

(iii) Solids. Operation of the Fluorinel process will result in increased production of two types of radioactive solids. (1) The process will filter undissolved solids from the dissolver product liquids. These solids will contain a portion of the fuel element isotopic spectrum, including transuranic waste. The annual volume of transuranic solid waste is estimated to be less than one cubic meter. This waste will be deposited in "filtered solids" containers and placed in a shielded 30-gallon drum. The drum will be placed in a shielded transfer cask and transferred to the INEL Radioactive Waste Management Complex (RWMC) for storage with other transuranic waste until permanent disposition can be made. (2) Spent fuel hangers, used to lower fuel elements into the dissolver, will also constitute additional wastes. The hangers will be decontaminated as far as practical, cut as necessary, packaged, and transferred to the INEL RWMC. The annual volume of fuel hanger solids is estimated to be less than one cubic meter.

Production of other radioactive solids is not expected to exceed that of the current zirconium headend. The use of remote maintenance and operation requires equipment and materials that will last longer than equipment and materials in the present headend, which results in fewer radioactive components being removed from the process. Generation of routine radioactive solid waste (metal scraps, spent filters, contaminated rags, paper, trash, etc.) normally disposed of to the INEL RWMC will not be measurably increased by the operation of the Fluorinel process over that produced by existing processes.

Nonradiological solid waste associated with the Fluorinel process will be the same as for other chemical processing operations. The majority of this waste consists of shipping containers of various types, used cleaning materials, broken and expended tools and clothing items, and other similar materials. These items will be used in nonradiation areas or disposed of in the INEL Sanitary Landfill, as appropriate. Since the Fluorinel headend will replace the zirconium headend, the total annual amount of nonradioactive solid waste generated by ICPP is not expected to increase.

(2) Fuel Storage Effluents. The fuel storage facility will not release measurable amounts of radioactivity to the environment during routine operation. Underwater storage and handling of fuel elements and the canning of any elements as necessary will preclude the release of measurable amounts of radioactivity during normal operations.

The potential impact on the environment of radioactive airborne, liquid, or solid effluents resulting from the operation of the fuel storage facility will be negligible. These effluents are discussed in the following subsections.

(i) Airborne. Normal airborne releases from the facility will be limited to radioactive noble gases and some radioiodine which may be released from the stored elements, or which may be released during underwater cutting operations. These releases, if they occur at all, will be at concentrations, at the INEL boundary, not greater than 3×10^{-3} percent of ERDAM 0524 uncontrolled area limits (see Table E-8). Airborne radioactive materials released to the building atmosphere from the pool water will be passed through prefilters and multiple HEPA filters, prior to release to the environment via the new 50-meter-stack.

TABLE E-9

ANNUAL BOUNDARY CONCENTRATIONS AND DOSES RESULTING FROM RELEASES FROM THE FUEL STORAGE FACILITY DURING NORMAL OPERATION.

Nuclide[a]	Average Annual Boundary Concentraction (μCi/ml)	% of ERDAM 0524 Uncontrolled Area RCG	Annual Whole Body Dose from Immersion and Inhalation (mRem)[c]
^{3}H	3E-13[b]	1.4E-4	2.7E-4
^{85}Kr	8E-12	2.8E-3	1.8E-2
^{90}Sr	2E-25	7E-13	1.2E-5
^{106}Ru	6E-26	3E-14	1E-15
^{106}Rh	6E-26	6E-14	9E-16
^{129}I	1E-19	6E-7	6E-9
^{137}Cs	1E-20	2E-9	3E-9
^{238}Pu	9E-29	1E-13	1E-13
^{239}Pu	2E-29	3E-14	3E-14
^{240}Pu	1E-29	2E-14	2E-14
^{241}Pu	3E-27	9E-12	6E-14

[a] Nuclides not listed have concentrations $< 2 \times 10^{-9}$% of RCG.

[b] 3R-13 means 3×10^{-13}.

[c] Calculated using appropriate conversion factors listed in A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment, ORNL-4992, Oak Ridge National Laboratory, Oak Ridge, Tennessee (March 1976).

The estimated annual whole body dose to an individual at the ne t point or the site boundary is 0.018 man-rem, which is 3.7×10^{-3} pe of the ERDAM 0524 Radiation Protection Standard for individuals in uncontrolled areas and is equivalent to about 0.01 percent of the annual exposure one would receive from the natural radioactivity background at INEL. Thus, the anticipated fuel storage airborne releases are expected to have a negligible effect on the environment.

(ii) Liquids. All liquid releases from the facility will be to existing ICPP liquid waste systems. Releases will include both nonradioactive and radioactive liquid wastes.

Nonradioactive liquid waste will include sewage and service wastes. The sewage will be transferred to the existing sewage treatment plant (CPP-715). Service waste (consisting of steam condensate, cooling water, liquid from nonradioactive drains, and other liquid waste with no radioactive contamination) will be transferred to the ICPP Service Waste System, monitored for radioactivity and sampled for chemical composition, and discharged to the ground via the existing disposal well (CPP-304). The disposal well is 182.3 meters deep and 25.4 cm in diameter and penetrates 42.7 meters into the aquifer. Service waste is discharged at ambient temperature and, therefore, has no thermal impact on the environment.

Radioactively contaminated liquid waste will be transferred through double-walled piping to a hold tank for monitoring. Vessel WL-104, adjacent to CPP-641, will receive this liquid, which will be transferred to the waste evaporator, located at CPP-604, on a batch basis.

Radioactively contaminated liquid wastes are concentrated and combined with other radioactive liquid wastes in underground storage tanks. These wastes are processed though the waste calciner, and the solids produced are placed in underground storage bins contained in reinforced concrete vaults located within the ICPP boundary.

The only potential chemical waste produced at the storage facility will be very small amounts of used cask decontamination solutions. These solutions will be treated as radioactively contaminated liquid wastes and disposed of as previously discussed.

The present ICPP liquid waste systems have sufficient capacity to handle all liquid releases from the fuel storage facility.

The impact of liquid waste discharge on the environment will be negligible.

(iii) Solids. Generation of routine radioactive solid waste (metal scraps, spend HEPA filters, spent ion exchange resin, contaminated rags, paper, trash, etc.), normally disposed of to the INEL Radioactive Waste Management Complex (RWMC) will not be measurably increased by the operation of the fuel storage facility over that produced by existing processes.

During construction, some solid waste may be generated by the removal of abandoned underground piping or contaminated soil. Piping will be decontaminated as necessary and packaged, and the soil will be packaged, and transferred to the RWMC. Piping and contaminated soil would eventually be removed during decommissioning of the ICPP and, therefore, are not "additional" solid waste.

Generation of nonradioactive solid waste will be the same as for the present fuel storage facility (CPP-603). Since the new facility replaces the present facility, the net increase in the rate of generation of nonradiological solid waste will be zero.

(3) Overall Facility. Operation of the facility will require electrical energy, chemicals, and other items normal to chemical processing and storage operations.

The primary benefit of operation of the Fluorinel process is the recovery of \$377 million worth of ^{235}U during the first eight years of operation.

The fuel storage facility will provide beneficial effects by serving as an improved storage area.

B. Utilities Replacement and Expansion, ICPP, Idaho

This project provides for the replacement of aging and deteriorated utilities (electrical, water, etc.) and expansion of existing utility production and distribution systems and tunnels to meet plant operating requirements. Equipment included in these systems are deepwell pumps, water demineralizers, emergency power generators, and electricl transformers and switchgear.

The Idaho Chemical Processing Plant was built in 1951 and has undergone a series of expansions and modifications. The originally installed utility systems have deteriorated, particularly where buried pipe and conduit are involved, to a point that the systems are unreliable. There is no spare capacity to serve plant growth. This project is necessary to replace these deteriorated systems and provide reliable, efficient systems to serve a multipurpose fuel reprocessing plant. When these utilities are replaced, added capacity will be built into each system to provide expansion for planned plant requirements and nominal reserve capacity.

This project will not involve the processing or handling of radioactive, explosive or toxic materials. It will be designed to minimize occupational or public health and safety hazards with no determental effect on human, health, safety welfare, or well being. It will not degrade water, air or land resources, nor will it affect ecological systems. Water pollution will be generated from existing boiler blowdown systems, water treatment facilities and sewage disposal facilities. A facility will be installed for removing the contaminants from the chemical wastes and aeration units will be installed for sanitary sewage treatment. These waste treatment systems will meet the requirements of Executive Order 11752.

C. Steam Generation, ICPP, INEL, Idaho

This project provides for the design and the eventual constructi of a coal-fired steam generation facility to meet plant operating requirements. The design shall include: (a) facilities for coal receiving, handling, and storage; (b) coal-fired boilers; (c) water treatment; (d) pollution control equipment; and (e) a building to house the necessary equipment. The total facility shall be designed to produce steam up to a maximum of 172,500 lbs/hr.

The Idaho Chemical Processing Plant was built in 1951 and has undergone a series of expansions and modifications. Some of the installed steam generation equipment has deteriorated to a point that the equipment is unreliable. There is no spare capacity to serve plant growth. This project is necessary to provide for the design of a new steam generation facility which will replace, in part, the existing oil-fired equipment with coal-fired equipment and will provide added capacity to provide expansion for planned plant requirements and normal reserve capacity.

This project will not involve the processing or handling of radioactive, explosive or toxic materials. It will be designed to minimize occupational or public health and safety hazards with no determental effect on human, health, safety welfare, or well being. It will not degrade water, air or land resources, nor will it affect ecological systems. Air pollution will be generated from the boiler exhaust, but pollution will be held to a minimum with pollution control equipment to be installed as part of this project. Water pollution will be generat from boiler blowdown and water treatment systems. A facility will be installed for removing the contaminants from these wastes. The waste treatment systems will meet the requirements of Executive Order 11752.

D. Personnel Protection and Support Facility, ICPP, INEL, Idaho

This project provides for modifications to the Idaho Chemical Processing Plant (ICPP) for reduction of radiation exposure and upgrading of plant maintenance capability and safety of operations to a level consistent with ERDA and other applicable safety standards.

The reduction of radiation exposure will be accomplished principally by (a) modifying certain cell hatches to permit television viewing, remote access for inspection, radiation surveys and decontamination; (b) modifying the existing sampling system in order to better contain the contamination and provide shielding for radiation; and (c) relocating the centrifuge which clarifies the feed stream to the first-cycle solvent extraction system, from an operating area to a processing cell, and providing for semiremote decontamination in the event of failure.

Other modifications which will be made to reduce radiation exposure and decrease radioactive decontamination are electrolytic cell upgrade, lining cell entrances with stainless steel, remote HEPA filter monitoring, vessel modification, and improved sampler vent system.

The upgrading of plant maintenance capability and safety will be accomplished by constructing a new building having approximately 50,000 square feet (including 15,000 square feet on mezzanine floor) of floor space.

The proposed building will be located within the perimeter of the ICPP and will require no significant improvements to land. Utilities such as electricity, water, steam, and sewage disposal will be provided by connection to existing systems.

Renovationed space used for existing maintenance functions will be used for a pilot plant office.

The purpose of this project is to reduce radiation exposure to personnel at the ICPP by upgrading facilities in the main processing building (Building CPP 601), and to provide sufficient safe working space for expanding maintenance activities.

The ICPP has been in operation for more than 20 years and is designed for direct (vs. remote) maintenance, dictating that process equipment and working areas be thoroughly decontaminated before maintenance, dictating that process equipment and working areas be thoroughly decontaminated before maintenance and inspection work can be undertaken.

Total radiation exposures at the ICPP have doubled during the past two years. The increase, which without remedial action can be expected to continue, is attributable to increasing maintenance requirements in an aging plant, and to increasing operational requirements resulting from an expanding program. The remedial actions proposed in this project will lower personnel radiation exposures to comply with ERDA Manual Chapter 9524 (dated April 8, 1975), which specifies a reduction in permissible skin exposure from 30 rem/year to 15 rem/year, and reduce total ICPP radiation exposures by about half of current levels (down to $\sim$150-200 man-rem).

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